TABLE 1—CHEMISTRY FACTOR FOR WELD METALS, °F—Continued

Copper, wt-%	Nickel, wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE 2—CHEMISTRY FACTOR FOR BASE METALS, °F

Copper, wt-%	Nickel, wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	208	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239

TABLE 2—CHEMISTRY FACTOR FOR BASE METALS, °F—Continued

Copper, wt-%	Nickel, wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

[60 FR 65468, Dec. 19, 1995, as amended at 61 FR 39300, July 29, 1996; 72 FR 49500, Aug. 28, 2007; 73 FR 5722, Jan. 31, 2008; 75 FR 23, Jan. 4, 2010]

§ 50.61a Alternate fracture toughness requirements for protection against pressurized thermal shock events.

(a) *Definitions.* Terms in this section have the same meaning as those presented in 10 CFR 50.61(a), with the exception of the term ''ASME Code.''

(1) *ASME Code* means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division I, ''Rules for the Construction of Nuclear Power Plant Components,'' and Section XI, Division I, ''Rules for Inservice Inspection of Nuclear Power Plant Components,'' edition and addenda and any limitations and modifications thereof as specified in § 50.55a.

(2) RT_{MAX-AW} means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found along axial weld fusion lines. RT_{MAX-AW} is determined under the provisions of paragraph (f) of this section and has units of °F.

(3) RT_{MAX-PL} means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found in plates in regions that are not associated with welds found in plates. RT_{MAX-PL} is determined under the provisions of paragraph (f) of this section and has units of °F.

(4) RT_{MAX-FO} means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws in forgings that are not associated with welds found in forgings. RT_{MAX-FO} is determined under

the provisions of paragraph (f) of this section and has units of °F.

(5) RT_{MAX-CW} means the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found along the circumferential weld fusion lines. RT_{MAX-CW} is determined under the provisions of paragraph (f) of this section and has units of °F.

(6) RT_{MAX-X} means any or all of the material properties RT_{MAX-AW}, RT_{MAX-PL}, RT_{MAX-FO}, RT_{MAX-CW}, or sum of RT_{MAX-AW} and RT_{MAX-PL}, for a particular reactor vessel.

(7) φt means fast neutron fluence for neutrons with energies greater than 1.0 MeV. φt is utilized under the provisions of paragraph (g) of this section and has units of n/cm².

(8) φ means average neutron flux for neutrons with energies greater than 1.0 MeV. φ is utilized under the provisions of paragraph (g) of this section and has units of n/cm²/sec.

(9) ΔT_{30} means the shift in the Charpy V-notch transition temperature at the 30 ft-lb energy level produced by irradiation. The ΔT_{30} value is utilized under the provisions of paragraph (g) of this section and has units of °F.

(10) *Surveillance data* means any data that demonstrates the embrittlement trends for the beltline materials, including, but not limited to, surveillance programs at other plants with or without a surveillance program integrated under 10 CFR part 50, appendix H.

(11) T_C means cold leg temperature under normal full power operating conditions, as a time-weighted average from the start of full power operation through the end of licensed operation. T_C has units of °F.

(12) *CRP* means the copper rich precipitate term in the embrittlement model from this section. The CRP term is defined in paragraph (g) of this section.

(13) *MD* means the matrix damage term in the embrittlement model for this section. The MD term is defined in paragraph (g) of this section.

(b) *Applicability.* The requirements of this section apply to each holder of an operating license for a pressurized water nuclear power reactor whose construction permit was issued before February 3, 2010 and whose reactor vessel was designed and fabricated to the ASME Boiler and Pressure Vessel Code, 1998 Edition or earlier. The requirements of this section may be implemented as an alternative to the requirements of 10 CFR 50.61.

(c) *Request for approval.* Before the implementation of this section, each licensee shall submit a request for approval in the form of an application for a license amendment in accordance with § 50.90 together with the documentation required by paragraphs (c)(1), (c)(2), and (c)(3) of this section for review and approval by the Director of the Office of Nuclear Reactor Regulation (Director). The application must be submitted for review and approval by the Director at least three years before the limiting RT_{PTS} value calculated under 10 CFR 50.61 is projected to exceed the PTS screening criteria in 10 CFR 50.61 for plants licensed under this part.

(1) Each licensee shall have projected values of RT_{MAX-X} for each reactor vessel beltline material for the EOL fluence of the material. The assessment of RT_{MAX-X} values must use the calculation procedures given in paragraphs (f) and (g) of this section. The assessment must specify the bases for the projected value of RT_{MAX-X} for each reactor vessel beltline material, including the assumptions regarding future plant operation (e.g., core loading patterns, projected capacity factors); the copper (Cu), phosphorus (P), manganese (Mn), and nickel (Ni) contents; the reactor cold leg temperature (T_C); and the neutron flux and fluence values used in the calculation for each beltline material. Assessments performed under paragraphs (f)(6) and (f)(7) of this section, shall be submitted by the licensee to the Director in its license amendment application to utilize § 50.61a.

(2) Each licensee shall perform an examination and an assessment of flaws in the reactor vessel beltline as required by paragraph (e) of this section. The licensee shall verify that the requirements of paragraphs (e), (e)(1), (e)(2), and (e)(3) of this section have been met. The licensee must submit to the Director, in its application to use § 50.61a, the adjustments made to the

volumetric test data to account for NDE-related uncertainties as described in paragraph (e)(1) of this section, all information required by paragraph (e)(1)(iii) of this section, and, if applicable, analyses performed under paragraphs (e)(4), (e)(5) and (e)(6) of this section.

(3) Each licensee shall compare the projected RT_{MAX-X} values for plates, forgings, axial welds, and circumferential welds to the PTS screening criteria in Table 1 of this section, for the purpose of evaluating a reactor vessel's susceptibility to fracture due to a PTS event. If any of the projected RT_{MAX-X} values are greater than the PTS screening criteria in Table 1 of this section, then the licensee may propose the compensatory actions or plant-specific analyses as required in paragraphs (d)(3) through (d)(7) of this section, as applicable, to justify operation beyond the PTS screening criteria in Table 1 of this section.

(d) *Subsequent requirements*. Licensees who have been approved to use 10 CFR 50.61a under the requirements of paragraph (c) of this section shall comply with the requirements of this paragraph.

(1) Whenever there is a significant change in projected values of RT_{MAX-X}, so that the previous value, the current value, or both values, exceed the screening criteria before the expiration of the plant operating license; or upon the licensee's request for a change in the expiration date for operation of the facility; a re-assessment of RT_{MAX-X} values documented consistent with the requirements of paragraph (c)(1) and (c)(3) of this section must be submitted in the form of a license amendment for review and approval by the Director. If the surveillance data used to perform the re-assessment of RT_{MAX-X} values meet the requirements of paragraph (f)(6)(v) of this section, the licensee shall submit the data and the results of the analysis of the data to the Director for review and approval within one year after the capsule is withdrawn from the vessel. If the surveillance data meet the requirements of paragraph (f)(6)(vi) of this section, the licensee shall submit the data, the results of the analysis of the data, and proposed ΔT_{30} and RT_{MAX-X} values considering the surveillance data in the form of a license amendment to the Director for review and approval within two years after the capsule is withdrawn from the vessel. If the Director does not approve the assessment of RT_{MAX-X} values, then the licensee shall perform the actions required in paragraphs (d)(3) through (d)(7) of this section, as necessary, before operation beyond the PTS screening criteria in Table 1 of this section.

(2) The licensee shall verify that the requirements of paragraphs (e), (e)(1), (e)(2), and (e)(3) of this section have been met. The licensee must submit, within 120 days after completing a volumetric examination of reactor vessel beltline materials as required by ASME Code, Section XI, the adjustments made to the volumetric test data to account for NDE-related uncertainties as described in paragraph (e)(1) of this section and all information required by paragraph (e)(1)(iii) of this section in the form of a license amendment for review and approval by the Director. If a licensee is required to implement paragraphs (e)(4), (e)(5), and (e)(6) of this section, the information required in these paragraphs must be submitted in the form of a license amendment for review and approval by the Director within one year after completing a volumetric examination of reactor vessel materials as required by ASME Code, Section XI.

(3) If the value of RT_{MAX-X} is projected to exceed the PTS screening criteria, then the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criteria. The schedule for implementation of flux reduction measures may take into account the schedule for review and anticipated approval by the Director of detailed plant-specific analyses which demonstrate acceptable risk with RT_{MAX-X} values above the PTS screening criteria due to plant modifications, new information, or new analysis techniques.

(4) If the analysis required by paragraph (d)(3) of this section indicates that no reasonably practicable flux reduction program will prevent the RT_{MAX-X} value for one or more reactor

vessel beltline materials from exceeding the PTS screening criteria, then the licensee shall perform a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent the potential for an unacceptably high probability of failure of the reactor vessel as a result of postulated PTS events. In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis and the description of the modifications must be submitted to the Director in the form of a license amendment at least three years before RT_{MAX-X} is projected to exceed the PTS screening criteria.

(5) After consideration of the licensee's analyses, including effects of proposed corrective actions, if any, submitted under paragraphs (d)(3) and (d)(4) of this section, the Director may, on a case-by-case basis, approve operation of the facility with RT_{MAX-X} values in excess of the PTS screening criteria. The Director will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision. The Director shall impose the modifications to equipment, systems and operations described to meet paragraph (d)(4) of this section.

(6) If the Director concludes, under paragraph (d)(5) of this section, that operation of the facility with RT_{MAX-X} values in excess of the PTS screening criteria cannot be approved on the basis of the licensee's analyses submitted under paragraphs (d)(3) and (d)(4) of this section, then the licensee shall request a license amendment, and receive approval by the Director, before any operation beyond the PTS screening criteria. The request must be based on modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or on further analyses based on new information or improved methodology. The licensee must show that the proposed alternatives provide reasonable assurance of adequate protection of the public health and safety.

(7) If the limiting RT_{MAX-X} value of the facility is projected to exceed the PTS screening criteria and the requirements of paragraphs (d)(3) through (d)(6) of this section cannot be satisfied, the reactor vessel beltline may be given a thermal annealing treatment under the requirements of § 50.66 to recover the fracture toughness of the material. The reactor vessel may be used only for that service period within which the predicted fracture toughness of the reactor vessel beltline materials satisfy the requirements of paragraphs (d)(1) through (d)(6) of this section, with RT_{MAX-X} values accounting for the effects of annealing and subsequent irradiation.

(e) *Examination and flaw assessment requirements.* The volumetric examination results evaluated under paragraphs (e)(1), (e)(2), and (e)(3) of this section must be acquired using procedures, equipment and personnel that have been qualified under the ASME Code, Section XI, Appendix VIII, Supplement 4 and Supplement 6, as specified in 10 CFR 50.55a(b)(2)(xv).

(1) The licensee shall verify that the flaw density and size distributions within the volume described in ASME Code, Section XI,[1] Figures IWB–2500–1 and IWB–2500–2 and limited to a depth from the clad-to-base metal interface of 1-inch or 10 percent of the vessel thickness, whichever is greater, do not exceed the limits in Tables 2 and 3 of this section based on the test results from the volumetric examination. The values in Tables 2 and 3 represent actual flaw sizes. Test results from the volumetric examination may be adjusted to account for the effects of NDE-related uncertainties. The methodology to account for NDE-related uncertainties must be based on statistical data from the qualification tests and any other tests that measure the difference between the actual flaw size and the NDE detected flaw size. Licensees who adjust their test data to account for NDE-related uncertainties to

[1] For forgings susceptible to underclad cracking the determination of the flaw density for that forging from the licensee's inspection shall exclude those indications identified as underclad cracks.

verify conformance with the values in Tables 2 and 3 shall prepare and submit the methodology used to estimate the NDE uncertainty, the statistical data used to adjust the test data and an explanation of how the data was analyzed for review and approval by the Director in accordance with paragraphs (c)(2) and (d)(2) of this section. The verification of the flaw density and size distributions shall be performed line-by-line for Tables 2 and 3. If the flaw density and size distribution exceeds the limitations specified in Tables 2 and 3 of this section, the licensee shall perform the analyses required by paragraph (e)(4) of this section. If analyses are required in accordance with paragraph (e)(4) of this section, the licensee must address the effects on through-wall crack frequency (TWCF) in accordance with paragraph (e)(5) of this section and must prepare and submit a neutron fluence map in accordance with the requirements of paragraph (e)(6) of this section.

(i) The licensee shall determine the allowable number of weld flaws in the reactor vessel beltline by multiplying the values in Table 2 of this section by the total length of the reactor vessel beltline welds that were volumetrically inspected and dividing by 1000 inches of weld length.

(ii) The licensee shall determine the allowable number of plate or forging flaws in their reactor vessel beltline by multiplying the values in Table 3 of this section by the total surface area of the reactor vessel beltline plates or forgings that were volumetrically inspected and dividing by 1000 square inches.

(iii) For each flaw detected in the inspection volume described in paragraph (e)(1) with a through-wall extent equal to or greater than 0.075 inches, the licensee shall document the dimensions of the flaw, including through-wall extent and length, whether the flaw is axial or circumferential in orientation and its location within the reactor vessel, including its azimuthal and axial positions and its depth embedded from the clad-to-base metal interface.

(2) The licensee shall identify, as part of the examination required by paragraph (c)(2) of this section and any subsequent ASME Code, Section XI ultrasonic examination of the beltline welds, any flaws within the inspection volume described in paragraph (e)(1) of this section that are equal to or greater than 0.075 inches in through-wall depth, axially-oriented, and located at the clad-to-base metal interface. The licensee shall verify that these flaws do not open to the vessel inside surface using surface or visual examination technique capable of detecting and characterizing service induced cracking of the reactor vessel cladding.

(3) The licensee shall verify, as part of the examination required by paragraph (c)(2) of this section and any subsequent ASME Code, Section XI ultrasonic examination of the beltline welds, that all flaws between the clad-to-base metal interface and three-eighths of the reactor vessel thickness from the interior surface are within the allowable values in ASME Code, Section XI, Table IWB–3510–1.

(4) The licensee shall perform analyses to demonstrate that the reactor vessel will have a TWCF of less than 1×10^{-6} per reactor year if the ASME Code, Section XI volumetric examination required by paragraph (c)(2) or (d)(2) of this section indicates any of the following:

(i) The flaw density and size in the inspection volume described in paragraph (e)(1) exceed the limits in Tables 2 or 3 of this section;

(ii) There are axial flaws that penetrate through the clad into the low alloy steel reactor vessel shell, at a depth equal to or greater than 0.075 inches in through-wall extent from the clad-to-base metal interface; or

(iii) Any flaws between the clad-to-base metal interface and three-eighths[2] of the vessel thickness exceed the size allowable in ASME Code, Section XI, Table IWB–3510–1.

(5) The analyses required by paragraph (e)(4) of this section must address the effects on TWCF of the known sizes and locations of all flaws detected by the ASME Code, Section XI, Appendix VIII, Supplement 4 and

[2] Because flaws greater than three-eights of the vessel wall thickness from the inside surface do not contribute to TWCF, flaws greater than three-eights of the vessel wall thickness from the inside surface need not be analyzed for their contribution to PTS.

Supplement 6 ultrasonic examination out to three-eighths of the vessel thickness from the inner surface, and may also take into account other reactor vessel-specific information, including fracture toughness information.

(6) For all flaw assessments performed in accordance with paragraph (e)(4) of this section, the licensee shall prepare and submit a neutron fluence map, projected to the date of license expiration, for the reactor vessel beltline clad-to-base metal interface and indexed in a manner that allows the determination of the neutron fluence at the location of the detected flaws.

(f) *Calculation of RT_{MAX-X} values*. Each licensee shall calculate RT_{MAX-X} values for each reactor vessel beltline material using φt. The neutron flux ($\varphi[t]$), must be calculated using a methodology that has been benchmarked to experimental measurements and with quantified uncertainties and possible biases.[3]

(1) The values of RT_{MAX-AW}, RT_{MAX-PL}, RT_{MAX-FO}, and RT_{MAX-CW} must be determined using Equations 1 through 4 of this section. When calculating RT_{MAX-AW} using Equation 1, RT_{MAX-AW} is the maximum value of ($RT_{NDT(U)} + \Delta T_{30}$) for the weld and for the adjoining plates. When calculating RT_{MAX-CW} using Equation 4, RT_{MAX-CW} is the maximum value of ($RT_{NDT(U)} + \Delta T_{30}$) for the circumferential weld and for the adjoining plates or forgings.

(2) The values of ΔT_{30} must be determined using Equations 5, 6 and 7 of this section, unless the conditions specified in paragraph (f)(6)(v) of this section are not met, for each axial weld, plate, forging, and circumferential weld. The ΔT_{30} value for each axial weld calculated as specified by Equation 1 of this section must be calculated for the maximum fluence ($\varphi t_{AXIAL-WELD}$) occurring along a particular axial weld at the clad-to-base metal interface. The ΔT_{30} value for each plate calculated as specified by Equation 1 of this section must also be calculated using the same value of $\varphi t_{AXIAL-WELD}$ used for the axial weld. The ΔT_{30} values in Equation 1 shall be calculated for the weld itself and each adjoining plate. The ΔT_{30} value for each plate or forging calculated as specified by Equations 2 and 3 of this section must be calculated for the maximum fluence (φt_{MAX}) occurring at the clad-to-base metal interface over the entire area of each plate or forging. In Equation 4, the fluence ($\varphi t_{WELD-CIRC}$) value used for calculating the plate, forging, and circumferential weld ΔT_{30} value is the maximum fluence occurring for each material along the circumferential weld at the clad-to-base metal interface. The ΔT_{30} values in Equation 4 shall be calculated for the circumferential weld and for the adjoining plates or forgings. If the conditions specified in paragraph (f)(6)(v) of this section are not met, licensees must propose ΔT_{30} and RT_{MAX-X} values in accordance with paragraph (f)(6)(vi) of this section.

(3) The values of Cu, Mn, P, and Ni in Equations 6 and 7 of this section must represent the best estimate values for the material. For a plate or forging, the best estimate value is normally the mean of the measured values for that plate or forging. For a weld, the best estimate value is normally the mean of the measured values for a weld deposit made using the same weld wire heat number as the critical vessel weld. If these values are not available, either the upper limiting values given in the material specifications to which the vessel material was fabricated, or conservative estimates (*i.e.*, mean plus one standard deviation) based on generic data[4] as shown in Table 4 of this section for P and Mn, must be used.

(4) The values of $RT_{NDT(U)}$ must be evaluated according to the procedures in the ASME Code, Section III, paragraph NB–2331. If any other method is used for this evaluation, the licensee shall submit the proposed method for review and approval by the Director along with the calculation of RT_{MAX-X} values required in paragraph (c)(1) of this section.

(i) If a measured value of $RT_{NDT(U)}$ is not available, a generic mean value of

[3] Regulatory Guide 1.190 dated March 2001, establishes acceptable methods for determining neutron flux.

[4] Data from reactor vessels fabricated to the same material specification in the same shop as the vessel in question and in the same time is an example of ''generic data.''

$RT_{NDT(U)}$ for the class[5] of material must be used if there are sufficient test results to establish a mean.

(ii) The following generic mean values of $RT_{NDT(U)}$ must be used unless justification for different values is provided: 0 °F for welds made with Linde 80 weld flux; and −56 °F for welds made with Linde 0091, 1092, and 124 and ARCOS B–5 weld fluxes.

(5) The value of T_C in Equation 6 of this section must represent the time-weighted average of the reactor cold leg temperature under normal operating full power conditions from the beginning of full power operation through the end of licensed operation.

(6) The licensee shall verify that an appropriate RT_{MAX-X} value has been calculated for each reactor vessel beltline material by considering plant-specific information that could affect the use of the model (*i.e.*, Equations 5, 6 and 7) of this section for the determination of a material's ΔT_{30} value.

(i) The licensee shall evaluate the results from a plant-specific or integrated surveillance program if the surveillance data satisfy the criteria described in paragraphs (f)(6)(i)(A) and (f)(6)(i)(B) of this section:

(A) The surveillance material must be a heat-specific match for one or more of the materials for which RT_{MAX-X} is being calculated. The 30-foot-pound transition temperature must be determined as specified by the requirements of 10 CFR part 50, Appendix H.

(B) If three or more surveillance data points measured at three or more different neutron fluences exist for a specific material, the licensee shall determine if the surveillance data show a significantly different trend than the embrittlement model predicts. This must be achieved by evaluating the surveillance data for consistency with the embrittlement model by following the procedures specified by paragraphs (f)(6)(ii), (f)(6)(iii), and (f)(6)(iv) of this section. If fewer than three surveillance data points exist for a specific material, then the embrittlement model must be used without performing the consistency check.

(ii) The licensee shall estimate the mean deviation from the embrittlement model for the specific data set (*i.e.*, a group of surveillance data points representative of a given material). The mean deviation from the embrittlement model for a given data set must be calculated using Equations 8 and 9 of this section. The mean deviation for the data set must be compared to the maximum heat-average residual given in Table 5 or derived using Equation 10 of this section. The maximum heat-average residual is based on the material group into which the surveillance material falls and the number of surveillance data points. For surveillance data sets with greater than 8 data points, the maximum credible heat-average residual must be calculated using Equation 10 of this section. The value of σ used in Equation 10 of this section must be obtained from Table 5 of this section.

(iii) The licensee shall estimate the slope of the embrittlement model residuals (estimated using Equation 8) plotted as a function of the base 10 logarithm of neutron fluence for the specific data set. The licensee shall estimate the T-statistic for this slope (T_{SURV}) using Equation 11 and compare this value to the maximum permissible T-statistic (T_{MAX}) in Table 6. For surveillance data sets with greater than 15 data points, the T_{MAX} value must be calculated using Student's T distribution with a significance level (α) of 1 percent for a one-tailed test.

(iv) The licensee shall estimate the two largest positive deviations (*i.e.*, outliers) from the embrittlement model for the specific data set using Equations 8 and 12. The licensee shall compare the largest normalized residual (r^*) to the appropriate allowable value from the third column in Table 7 and the second largest normalized residual to the appropriate allowable value from the second column in Table 7.

(v) The ΔT_{30} value must be determined using Equations 5, 6, and 7 of this section if all three of the following criteria are satisfied:

(A) The mean deviation from the embrittlement model for the data set is

[5] The class of material for estimating $RT_{NDT(U)}$ must be determined by the type of welding flux (Linde 80, or other) for welds or by the material specification for base metal.

equal to or less than the value in Table 5 or the value derived using Equation 10 of this section;

(B) The T-statistic for the slope (T_{SURV}) estimated using Equation 11 is equal to or less than the Maximum permissible T-statistic (T_{MAX}) in Table 6; and

(C) The largest normalized residual value is equal to or less than the appropriate allowable value from the third column in Table 7 and the second largest normalized residual value is equal to or less than the appropriate allowable value from the second column in Table 7. If any of these criteria is not satisfied, the licensee must propose ΔT_{30} and RT_{MAX-X} values in accordance with paragraph (f)(6)(vi) of this section.

(vi) If any of the criteria described in paragraph (f)(6)(v) of this section are not satisfied, the licensee shall review the data base for that heat in detail, including all parameters used in Equations 5, 6, and 7 of this section and the data used to determine the baseline Charpy V-notch curve for the material in an unirradiated condition. The licensee shall submit an evaluation of the surveillance data to the NRC and shall propose ΔT_{30} and RT_{MAX-X} values, considering their plant-specific surveillance data, to be used for evaluation relative to the acceptance criteria of this rule. These evaluations must be submitted for review and approval by the Director in the form of a license amendment in accordance with the requirements of paragraphs (c)(1) and (d)(1) of this section.

(7) The licensee shall report any information that significantly influences the RT_{MAX-X} value to the Director in accordance with the requirements of paragraphs (c)(1) and (d)(1) of this section.

(g) *Equations and variables used in this section.*

$$\text{Equation 1: } RT_{MAX-AW} = MAX\left\{\left[RT_{NDT(U)-plate} + \Delta T_{30-plate}\right], \left[RT_{NDT(U)-axial\ weld} + \Delta T_{30-axial\ weld}\right]\right\}$$

$$\text{Equation 2: } RT_{MAX-PL} = RT_{NDT(U)-plate} + \Delta T_{30-plate}$$

$$\text{Equation 3: } RT_{MAX-FO} = RT_{NDT(U)-forging} + \Delta T_{30-forging}$$

$$\text{Equation 4: } RT_{MAX-CW} = MAX\left\{\left[RT_{NDT(U)-plate} + \Delta T_{30-plate}\right], \left[RT_{NDT(U)-circweld} + \Delta T_{30-circweld}\right], \left[RT_{NDT(U)-forging} + \Delta T_{30-forging}\right]\right\}$$

$$\text{Equation 5: } \Delta T_{30} = MD + CRP$$

$$\text{Equation 6: } MD = A \times (1 - 0.001718 \times T_C) \times \left(1 + 6.13 \times P \times Mn^{2.471}\right) \times \varphi t_e^{0.5}$$

$$\text{Equation 7: } CRP = B \times \left(1 + 3.77 \times Ni^{1.191}\right) \times f(Cu_e, P) \times g(Cu_e, Ni, \varphi t_e)$$

Where:

P [wt-&%] = phosphorus content
Mn [wt-%] = manganese content
Ni [wt-%] = nickel content
Cu [wt-%] = copper content
A = 1.140×10^{-7} for forgings
A = 1.561×10^{-7} for plates
A = 1.417×10^{-7} for welds
B = 102.3 for forgings
B = 102.5 for plates in non-Combustion Engineering manufactured vessels
B = 135.2 for plates in Combustion Engineering vessels
B = 155.0 for welds

$\varphi t_e = \varphi t$ for $\varphi \geq 4.39 \times 10^{10}$ n/cm²/sec
$\varphi t_e = \varphi t \times (4.39 \times 10^{10}/\varphi)^{0.2595}$ for $\varphi < 4.39 \times 10^{10}$ n/cm²/sec

Where:

φ [n/cm²/sec] = average neutron flux
t [sec] = time that the reactor has been in full power operation
φt [n/cm²] = $\varphi \times t$

$f(Cu_e,P) = 0$ for $Cu \leq 0.072$

$f(Cu_e,P) = [Cu_e - 0.072]^{0.668}$ for $Cu > 0.072$ and $P \leq 0.008$

$f(Cu_e,P) = [Cu_e - 0.072 + 1.359 \times (P - 0.008)]^{0.668}$ for $Cu > 0.072$ and $P > 0.008$

Where:

$Cu_e = 0$ for $Cu \leq 0.072$

$Cu_e = MIN$ (Cu, maximum Cu_e) for $Cu > 0.072$

maximum $Cu_e = 0.243$ for Linde 80 welds

maximum $Cu_e = 0.301$ for all other materials

$g(Cu_e,Ni,\varphi t_e) = 0.5 + (0.5 \times \tanh\{[\log_{10}(\varphi t_e) + (1.1390 \times Cu_e) - (0.448 \times Ni) - 18.120]/0.629\}$

Equation 8: Residual (r) = measured ΔT_{30} − predicted ΔT_{30} (by Equations 5, 6 and 7)

$$\text{Equation 9: Mean deviation for a data set of n data points} = (1/n) \times \sum_{i=1}^{n} r_i$$

Equation 10: Maximum credible heat-average residual = $2.33\sigma/n^{0.5}$

Where:

n = number of surveillance data points (sample size) in the specific data set

σ = standard deviation of the residuals about the model for a relevant material group given in Table 5.

$$\text{Equation 11: } T_{SURV} = \frac{m}{se(m)}$$

Where:

m is the slope of a plot of all of the r values (estimated using Equation 8) versus the base 10 logarithm of the neutron fluence for each r value. The slope shall be estimated using the method of least squares.

se(m) is the least squares estimate of the standard-error associated with the estimated slope value *m*.

$$\text{Equation 12: } r^* = \frac{r}{\sigma}$$

Where:

r is defined using Equation 8 and σ is given in Table 5

TABLE 1—PTS SCREENING CRITERIA

Product form and RT_{MAX-X} values	RT_{MAX-X} limits [°F] for different vessel wall thicknesses[6] (T_{WALL})		
	$T_{WALL} \leq 9.5$ in.	9.5 in. $< T_{WALL} \leq 10.5$ in.	10.5 in. $< T_{WALL} \leq 11.5$ in.
Axial Weld—RT_{MAX-AW}	269	230	222
Plate—RT_{MAX-PL}	356	305	293
Forging without underclad cracks—RT_{MAX-FO}[7]	356	305	293
Axial Weld and Plate—RT_{MAX-AW} + RT_{MAX-PL}	538	476	445
Circumferential Weld—RT_{MAX-CW}[8]	312	277	269
Forging with underclad cracks—RT_{MAX-FO}[9]	246	241	239

[6] Wall thickness is the beltline wall thickness including the clad thickness.

[7] Forgings without underclad cracks apply to forgings for which no underclad cracks have been detected and that were fabricated in accordance with Regulatory Guide 1.43.

[8] RT_{PTS} limits contribute 1×10^{-8} per reactor year to the reactor vessel TWCF.

[9] Forgings with underclad cracks apply to forgings that have detected underclad cracking or were not fabricated in accordance with Regulatory Guide 1.43.

TABLE 2—ALLOWABLE NUMBER OF FLAWS IN WELDS

Through-wall extent, TWE [in.] TWE_{MIN}	Through-wall extent, TWE [in.] TWE_{MAX}	Maximum number of flaws per 1,000-inches of weld length in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}
0	0.075	No Limit.
0.075	0.475	166.70.
0.125	0.475	90.80.
0.175	0.475	22.82.
0.225	0.475	8.66.
0.275	0.475	4.01.
0.325	0.475	3.01.
0.375	0.475	1.49.
0.425	0.475	1.00.
0.475	Infinite	0.00.

TABLE 3—ALLOWABLE NUMBER OF FLAWS IN PLATES AND FORGINGS

Through-wall extent, TWE [in.] TWE_{MIN}	Through-wall extent, TWE [in.] TWE_{MAX}	Maximum number of flaws per 1,000 square-inches of inside surface area in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}. This flaw density does not include underclad cracks in forgings
0	0.075	No Limit.
0.075	0.375	8.05.
0.125	0.375	3.15.
0.175	0.375	0.85.
0.225	0.375	0.29.
0.275	0.375	0.08.
0.325	0.375	0.01.
0.375	Infinite	0.00.

TABLE 4—CONSERVATIVE ESTIMATES FOR CHEMICAL ELEMENT WEIGHT PERCENTAGES

Materials	P	Mn
Plates	0.014	1.45
Forgings	0.016	1.11
Welds	0.019	1.63

TABLE 5—MAXIMUM HEAT-AVERAGE RESIDUAL [°F] FOR RELEVANT MATERIAL GROUPS BY NUMBER OF AVAILABLE DATA POINTS (SIGNIFICANCE LEVEL = 1%)

Material group	σ [°F]	Number of available data points					
		3	4	5	6	7	8
Welds, for Cu >0.072	26.4	35.5	30.8	27.5	25.1	23.2	21.7
Plates, for Cu >0.072	21.2	28.5	24.7	22.1	20.2	18.7	17.5
Forgings, for Cu >0.072	19.6	26.4	22.8	20.4	18.6	17.3	16.1
Weld, Plate or Forging, for Cu ≤0.072	18.6	25.0	21.7	19.4	17.7	16.4	15.3

TABLE 6—T_{MAX} VALUES FOR THE SLOPE DEVIATION TEST (SIGNIFICANCE LEVEL = 1%)

Number of available data points (n)	T_{MAX}
3	31.82
4	6.96
5	4.54
6	3.75
7	3.36
8	3.14
9	3.00
10	2.90
11	2.82
12	2.76
13	2.72
14	2.68
15	2.65

TABLE 7—THRESHOLD VALUES FOR THE OUTLIER DEVIATION TEST (SIGNIFICANCE LEVEL = 1%)

Number of available data points (n)	Second largest allowable normalized residual value (r*)	Largest allowable normalized residual value (r*)
3	1.55	2.71
4	1.73	2.81
5	1.84	2.88
6	1.93	2.93
7	2.00	2.98
8	2.05	3.02
9	2.11	3.06
10	2.16	3.09
11	2.19	3.12
12	2.23	3.14
13	2.26	3.17
14	2.29	3.19
15	2.32	3.21

[75 FR 23, Jan. 4, 2010, as amended at 75 FR 5495, Feb. 3, 2010; 75 FR 10411, Mar. 8, 2010; 75 FR 72653, Nov. 26, 2010]

§ 50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

(a) *Applicability*. The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted.

(b) *Definition*. For purposes of this section, *Anticipated Transient Without Scram* (ATWS) means an anticipated operational occurrence as defined in appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of appendix A of this part.

(c) *Requirements*. (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

(4) Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must